Air Quality Permitting
Statement of Basis

May 2, 2007

Permit to Construct No. P-2007.0043
and
Tier I Administrative Amendment No. T1-060521

U.S. Department of Energy, Idaho Operations Office (DOE-ID)
Idaho National Laboratory (INL)
Materials and Fuels Complex (MFC)
Fuel Conditioning Facility (FCF)
Scoville, Idaho

Facility ID No. 011-00022

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Air Quality Division

PROPOSED PERMIT for PUBLIC COMMENT

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Acronyms, Units, and Chemical Nomenclatures

AFS	AIRS Facility Subsystem
AIRS	Aerometric Information Retrieval System
AQCR	Air Quality Control Region
ANSI	American National Standards Institute
ASTM	American Society for Testing and Materials
BEA	Battelle Energy Alliance, LLC
CEMS	continuous emission monitoring system
CFR	Code of Federal Regulations
CO	carbon monoxide
DEQ	Department of Environmental Quality
DOE-ID	U.S. Department of Energy, Idaho Operations Office
EBR-II	Experimental Breeder Reactor No. 2
EDE	effective dose equivalent
EPA	U.S. Environmental Protection Agency
FCF	Fuel Conditioning Facility
FEIS	Environmental Impact Statement
FFTF	Fast Flux Test Facility
HAPs	Hazardous Air Pollutants
HEPA	high efficiency particulate air (filter)
HRA	hot repair area
HRF	hot repair facility
IDAPA	a numbering designation for all administrative rules in Idaho promulgated in accordance with the Idaho Administrative Procedures Act
NOx	nitrogen oxides
MFC	Materials and Fuels Complex
mrem/yr	millirems per year
NESHAP	Nation Emission Standards for Hazardous Air Pollutants
PM_{10}	particulate matter with an aerodynamic diameter less than or equal to a nominal 10 micrometers
PSD	Prevention of Significant Deterioration
PTC	permit to construct
PTE	potential to emit
QAPjP	Quality Assurance Project Plan
RSFFF	Research Scale Fuel Fabrication Facility
Rules	Rules for the Control of Air Pollution in Idaho
scfm	standard cubic feet per minute
SIP	State Implementation Plan
SO_2	sulfur dioxide

<u>Radionuclide Isotopes</u>
<u>(in the same order as in Table 5.1)</u>

Am-241	americium-241
Ar-41	argon-41
Co-60	cobalt-60
Cs-137	cesium-137
Ba-137	barium-137
H-3	tritium (hydrogen-3)
I-129	iodine-129
Kr-85	krypton-85
Pu-238	plutonium-238
Pu-239	plutonium-239
Sb-125	antimony-125
Te-125m	tellurium-125
Sr-90	strontium-90
Y-90	yttrium-90

1. PURPOSE

The purpose for this memorandum is to satisfy the requirements of IDAPA 58.01.01.200, Rules for the Control of Air Pollution in Idaho, for issuing permits to construct.

2. FACILITY DESCRIPTION

The primary mission of the Materials and Fuels Complex (MFC) Fuel Conditioning Facility (FCF) is electrometallurgical treatment of sodium-bonded spent metallic nuclear fuel from EBR-II, FERMI-1, the Fast Flux Test Facility (FFTF), and smaller amounts of other sodium-bonded fuels. Both driver and blanket fuel will be processed. A diagram of the FCF layout is included in Appendix A.

Current plans also call for the installation of two new processing capabilities. The Research Scale Fuel Fabrication Facility (RSFFF) Metal Fuel Module is a project designed to demonstrate all aspects of remote metal fuel fabrication technology on a small scale. Installation of oxide fuel fabrication capability is also planned to provide a backup capability for a demonstration project proposed to be undertaken at another National Laboratory.

3. FACILITY / AREA CLASSIFICATION

The MFC FCF is a facility located within the INL site. The INL is defined as a major facility for purposes of the Title V Program per IDAPA 58.01.01.008.10 because the facility has a potential to emit (PTE) of over 100 T/yr of a regulated air pollutant. For purposes of the PSD Program, the INL is classified as a “major stationary source” per IDAPA 58.01.01.205 [40 CFR 52.21b(1)] since it has the potential to emit over 250 T/yr of two regulated NSR pollutants (NO_x and CO), and it is not on the list of designated facilities. The AIRS classification is “A” because the facility has the PTE of over 100 T/yr of a regulated air pollutant.

This facility is located in Air Quality Control Region (AQCR) 61 and UTM Zone 12. It is located within the boundaries of the INL and Bingham County which is designated as unclassifiable for all regulated criteria pollutants (PM_{10}, CO, NO_x, SO_2, lead, and ozone).

The AIRS information for the “INL facility” is not changed as a result of this permit revision, therefore, a revised AIRS Facility Classification form is not included in this document.

4. APPLICATION SCOPE

The purpose of this Permit to Construct (PTC) is to:

- Add the following proposed new fuel fabrication capabilities:
  - Research Scale Fuel Fabrication Facility (RSFFF) Metal Fuel Module.
  - Oxide fuel fabrication demonstration capability.
- Delete the 30-day and annual disassembly and processing limits for fuel assemblies processed in the air and argon cells.
- Delete the 10 and 15 percent average heavy metal burnup requirement for spent fuels being treated.
- Revise facility and responsible official contacts for the MFC FCF.

4.1 Application Chronology

March 30, 2007	Receipt of PTC application and $1,000 application fee.
April 9, 2007	Application determined complete.
April 9, 2007	Draft permit and statement of basis sent for peer review and to the Idaho Falls Regional Office for review and comment.
April 13, 2007	Draft permit and statement of basis sent to the facility for review.
April 25, 2007	Receipt of minor comments from the facility.

5. PERMIT ANALYSIS

This section of the Statement of Basis describes the regulatory requirements for this PTC action.

5.1 Emissions Inventory

The spent fuel inventory is described in a 2000 Final Environmental Impact Statement (FEIS)[1] for sodium-bonded fuels from EBR-II, Fermi-1, FFTF, and from miscellaneous smaller sources (see Appendix A). Radiological emissions from electrometallurgical treatment at the FCF of the approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel in the DOE spent fuel inventory were also estimated in the FEIS. These estimates were based on processing 5,000 kilograms (11,023 pounds) of driver and blanket fuel each year over a 12-year project life, with a maximum of 600 kilograms (1,320 pounds) of heavy metal driver fuel processed in any year.

Radionuclide concentrations were based on the spent nuclear fuel with the highest representative radionuclide content with no credit taken for further decay beyond that which occurred prior to the year 2000. A summary of the fuel processing assumptions is shown in Table E-4 of the EIS, and annual radiological emissions (for a 12-year project life) and total emissions from processing all of the current inventory of sodium-bonded nuclear fuel are shown in Table E-5 of the FEIS. Copies of these two tables are included in Appendix A.

Radiological emissions from the MFC Main Stack are continuously monitored in accordance with 40 CFR 61.93. Stack emissions are reported to the EPA and Idaho DEQ in annual INL National Emission Standards for Hazardous Air Pollutants (NESHAP) reports, which must be submitted by June 30 of each year for the previous calendar year. A comparison of reported MFC Main Stack emissions for recent years[2] (when EBR-II driver fuel assemblies were being processed) to total emissions expected for processing the entire inventory is shown in Table 5.1. As shown in the table, the projected emissions are of a similar magnitude or greater than the actual emissions. Note, however, that the actual emissions of strontium-90 (Sr-90) and yttrium-90 (Y-90) have already exceeded the projected total Sr-90 and Y-90 emissions for the lifetime of the project.

[1] July 2000, U.S. Department of Energy, Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel, DOE/EIS-0306, accessed 4/12/2007, http://www.eh.doe.gov/nepa/eis/eis0306/eis0306.htm

[2] June 1999 – 2005, U.S. Department of Energy Idaho Operations Office, National Emission Standards for Hazardous Air Pollutants – Calendar Year [1999 through 2005] INEEL [INL] Report for Radionuclides.

Table 5.1 MFC MAIN STACK ANNUAL EMISSIONS COMPARED TO EIS TOTAL PROJECTIONS

	INL NESHAP Annual Reports – MFC Main Stack Emissions Source #MFC-764-001 (formerly ANL-764-001) (Curies)							DOE/EIS-0306 (Curies)	
Radionuclides Reported[a]	**1999**	**2000**	**2001**	**2002**	**2003**	**2004**	**2005**	**Years 1 - 5**	**Project Total**
Am-241								6.2E-12	3.3E-11
Ar-41									
Co-60								1.6E-09	8.8E-09
Cs-137/ Ba-137m								3.2E-08/ 4.0E-06	2.3E-05/ 2.2E-05
H-3	11.3	2.41	0.728	4.59	3.69	14.1	10.1	770	4,530
I-129								1.4E-12	8.2E-12
Kr-85	1,680	375	79.7	641	533	2,260	1,380	11,570	66,670
Pu-238								2.9E-10	1.6E-09
Pu-239								7.1E-09	3.7E-08
Sb-125/ Te-125m								4.1E-08/ 4.5E-10	2.4E-07/ 2.6E-07
Sr-90/ Y-90	---	---	4.23E-07	6.69E-07	5.41E-07	---	---	7.0E-08/ 7.0E-08	4.0E-07/ 4.0E-07
Xe-131m	8.82E-14	7.63E-22	1.06E-13	---	---	---	---	---	---

[a] Radionuclides reported from continuously compliance-monitored INL sources

The potential increase in emissions from the new demonstration fuel fabrication capabilities was not estimated. See the modeling section below regarding the estimated potential increase in the site dose equivalent from fuel fabrication activities.

5.2 Modeling

DOE/EIS-0306, Dose Equivalent Modeling for DOE Sodium-Bonded Nuclear Fuels

Modeling to estimate the dose equivalent that any member of the public might receive, in any year, from emissions from processing DOE's approximately 60-metric ton inventory of sodium-bonded spent nuclear fuel at MFC FCF over a 12-year period was conducted as part of preparing the 2000 Final EIS for treatment and management of the DOE inventory of sodium-bonded nuclear fuel.

Modeling in support of the EIS was conducted using the GENII computer model developed by Pacific Northwest National Laboratory. The GENII model is described in the EIS (pp. E-9 and E-10) as capable of analyzing environmental contamination resulting from acute or chronic releases to, or initial contamination in, air, water, or soil. Inhalation exposure for the maximally exposed offsite individual was assumed to be per year. The following assumptions were used:

- The actual 60.96-meter (200-foot) stack height was assumed to be the effective stack height (no plume rise was credited, which is a conservative assumption).
- Except for tritium (H-3), radionuclides were considered to be released in the chemical form resulting in the largest radiological impact. Electrometallurgical treatment of the fuels occurs in the argon cell, where the inert atmosphere is assumed to prevent oxidation of elemental tritium to tritium oxide.
- Krypton-85 and tritium were presumed to be released as gases with no capture in the HEPA filters.

The analysis conducted in support of the 2000 FEIS projected that the maximally exposed offsite individual would receive a dose of 3.4E-04 mrem/yr during project years 1 through 5, 2.8E-04 mrem/yr during project year 6, and 1.3E-07 mrem/yr during project years 7 through 12.

A summary of the dose that might be received by the maximally exposed offsite individual from processing driver and blanket fuels from different sources is shown in Table E-4 of the FEIS, and the cumulative maximum radiological impacts for a 12-year project life are shown in Table E-5 of that report. Copies of these two tables are included in Appendix A.

Previous Permitting Analysis, Dose Equivalent Estimate

The analysis for the offsite dose from processing EBR-II spent fuel (PTC No. 011-00022, issued December 5, 1989) projected that the maximally exposed offsite individual would receive a whole body dose of 3.62E-04 mrem/yr for each year that the facility was operated.

INL NESHAP Reports, Effective Dose Equivalents

As noted in the emission inventory discussion above, radionuclides emitted from sources at the INL are reported for each calendar year in an annual NESHAP report. The potential radiation dose from the emission sources is evaluated and modeled. Historically the effective dose equivalent (EDE) calculated for the INL NESHAP reports has always been between 0.01 and 0.1 mrem. In addition, historically, less than 5% of the radionuclide release sources at the INL cause the site dose to exceed 0.01 mrem. This category includes emissions from the MFC Main Stack. Sources that add less than 1.0E-05 mrem to the site dose are designated as “not significant contributors” because the values are small enough that the emissions from these sources do not affect the total calculated site dose.

The calculated INL EDEs reported in INL NESHAP reports for recent years are summarized in Table 5.2. For comparison, the annual dose limit for DOE operations at the INL, the estimated dose equivalents from the previous PTC analysis (for processing only EBR-II fuel at the FCF) and the dose equivalents estimated in the FEIS for processing the entire DOE inventory of sodium-bonded fuel at the MFC FCF are also included in Table 5.2.

Table 5.2 INL REPORTED EDE COMPARED TO PTC AND EIS DOSE PROJECTIONS

<table>
<tr><td></td><td colspan="7">Effective Dose Equivalent
(mrem)</td></tr>
<tr><td></td><td>1999</td><td>2000</td><td>2001</td><td>2002</td><td>2003</td><td>2004</td><td>2005</td></tr>
<tr><td>40 CFR 61, Subpart H, Annual Dose Limit for the DOE INL[a]</td><td>10</td><td>10</td><td>10</td><td>10</td><td>10</td><td>10</td><td>10</td></tr>
<tr><td>INL Site NESHAP Annual Reports, Modeled EDE (Entire Site) [a]</td><td>7.92E-03</td><td>3.40E-02</td><td>3.5E-02</td><td>5.50E-02</td><td>3.5E-02</td><td>4.4E-02</td><td>7.7E-02</td></tr>
<tr><td>INL Site NESHAP Annual Reports, Modeled EDE from the MCF Main Stack</td><td>2.18E-05</td><td>1.5E-05</td><td>NR [b]</td><td>NR</td><td>NR</td><td>NR</td><td>NR</td></tr>
<tr><td>MFC FCF - PTC No. 011-00022, 12/5/89
Predicted Dose Equivalent:
Presumed Processing of EBR-II Fuel only, presumed continued operation of the EBR-II facility.</td><td>3.62E-04</td><td>3.62E-04</td><td>3.62E-04</td><td>3.62E-04</td><td>3.62E-04</td><td>3.62E-04</td><td>3.62E-04</td></tr>
<tr><td></td><td>Years 1-5</td><td>Year 6</td><td>Years 7-12</td><td colspan="4">Project Total (Years 1-12)</td></tr>
<tr><td>MFC FCF - DOE/EIS-0306, July 2000
Predicted Dose Equivalent:
Presumed Processing of EBR-II fuel in addition to other sodium-bonded fuels in the DOE inventory.</td><td>3.4E-04 mrem/yr</td><td>2.8E-04 mrem/yr</td><td>1.3E-07 mrem/yr</td><td colspan="4">1.98E-03 mrem</td></tr>
</table>

[a] This represents the dose from all DOE activities at the INL. The contribution to the site dose from FCF activity is not always included in the annual NESHAP reports.

[b] Not reported.

Additional emissions that may occur from proposed metallic or oxide nuclear fuel fabrication demonstration capabilities at the FCF are not anticipated to be measurable when combined with fuel processing emissions in the MFC Main Stack. This is a reasonable assumption given that emissions from the MFC Fuel Manufacturing Facility stack (reported as Source #ANL-704-008 in INL NESHAP reports for 1999 through 2005) have consistently been determined not to significantly contribute to the total site dose, i.e., would contribute less than 1.0E-05 mrem to the annual site dose.

5.3 Regulatory Review

This section describes the regulatory analysis of the applicable air quality rules with respect to this PTC.

IDAPA 58.01.01.201..............................Permit to Construct Required

The facility has requested that several existing permit conditions be revised or deleted. Therefore, a PTC is required.

IDAPA 58.01.01.203..............................Permit Requirements for New and Modified Stationary Sources

The applicant has shown to the satisfaction of DEQ that the facility will comply with all applicable emissions standards, ambient air quality standards, and toxic increments. The only emissions of concern are radionuclides.

IDAPA 58.01.01.209.05.c.......................PTC Procedures for Tier I Sources

This PTC is for a Tier I source, therefore, the PTC may be processed according to the procedures for a Tier I source. The permittee may request that the PTC requirements be incorporated, at any time after issuance of the PTC, into the Tier I operating permit through an administrative amendment in accordance with IDAPA 381. The only change to the Tier I permit is to update the applicable PTC number and issuance date for the FCF. The Tier I permit is currently being administratively amended to incorporate another PTC, so this minor change will be incorporated as well. Based on this, the PTC and the Tier I administrative amendment will be processed concurrently. A draft PTC will be provided for public comment and affected state review per IDAPA 209.05.c, 364, and 365. The proposed PTC will also be sent to EPA for review per IDAPA 366.

IDAPA 58.01.01.210 Demonstration of Preconstruction Compliance with Toxic Standards

The applicant has demonstrated preconstruction compliance for all TAPs identified in the permit application. The only emissions of concern are radionuclides.

IDAPA 58.01.01.224..............................Permit to Construct Application Fee

The applicant satisfied the PTC application fee requirement by submitting a fee of $1,000.00 at the time the original application was submitted, March 30, 2007.

IDAPA 58.01.01.225..............................Permit to Construct Processing Fee

There is no change to the permitted total emissions from the proposed facility changes, but engineering analysis was required to develop appropriate alternate permit conditions. Therefore, the associated processing fee is $1,000.00. No permit to construct can be issued without first paying the required processing fee.

IDAPA 58.01.01.380, 381.................Changes to Tier I Operating Permits, Administrative Amendment

The Tier I operating permit will be changed as an Administrative amendment. Under IDAPA 58.01.01.381.01.e, the amendment is to incorporate into the Tier I operating permit the requirements from a PTC issued by the Department in accordance with Subsection 209.05.c.

IDAPA 58.01.01.591, 40 CFR 61 and 63 ....National Emissions Standards for Hazardous Air Pollutants (NESHAP)

The requirements of 40 CFR 61, Subpart H, National Emission Standards for Emissions of Radionuclides other than Radon from Department of Energy Facilities, apply to this project. These requirements are already specified for all emissions units at the INL in Section 2 (Facility-wide Conditions) of the INL Tier I operating permit. The requirement to continuously monitor the radionuclide emissions from the MFC Main Stack is included in Tier I Permit Condition 2.15.1, which requires that the permittee determine radionuclide emissions in accordance with 40 CFR 61.93. Compliance is demonstrated by meeting the requirements to continuously monitor the MFC Main Stack and by submitting the required annual NESHAP report for the INL.

The existing permit includes descriptions of emissions control systems that are installed and operated at the MFC FCF including the following: ventilation systems to maintain the building areas at negative pressures, and process and building ventilation HEPA filters. To ensure compliance with the NESHAP requirements, conditions were included in the permit to install, operate, maintain and monitor these systems. Since 40 CFR 61 Subpart H is not part of the Idaho State Implementation Plan (SIP), then the permit conditions established for the negative pressure ventilation systems and HEPA filters are “state-only requirements” which are included in the PTC but are not required to be included in the Tier I operating permit.

In addition, this permit action revises the method for monitoring material throughputs for this facility. The throughput limits inherently limit the radiological emissions, so are used as a surrogate to ensure compliance with the NESHAPs requirements. Like the HEPA conditions, the throughput limits are not included in the Tier I permit.

The installation of the fuel fabrication capabilities is not expected to result in a measurable increase of radiological emissions from the FCF. Based on the last annual NESHAP report, the facility is in compliance with 40 CFR Subpart H. In accordance with 40 CFR 61.96, it appears that an application for prior EPA approval under 40 CFR 61.08 or notification of startup under 61.09 is therefore not required.

The change in the allowable fuel processing limits and clarification with regard to the types of fuels allowed to be processed does not meet the definition of “construction” under 40 CFR 61 Subpart A.

5.4 Permit Conditions Review

This section describes only those permit conditions that have been created, revised, modified or deleted as a result of this permit action. All other permit conditions remain unchanged.

Tier I Condition 1.2

The only change to this condition is to replace “PTC No. 011-00022, issued May 9, 2001 for the FCF,” with “P-2007.0043” and its final issue date in the list of permits for the MFC.

PTC Condition 2.1

The process description was revised to include changes requested by the facility in the application.

PTC Conditions 2.3, 2.5, 2.10, 2.12, and Tier I Condition 2.15

The National Emissions Standards for Emissions of Radionuclides other than Radon from Department of Energy Facilities, 40 CFR 61 Subpart H, apply to the MFC FCF and, therefore, existing Conditions 2.1, 2.2, 4.2, and Appendix A Condition 1.1 regarding the emissions standards, operating, and monitoring were carried forward from the existing permit with only minor editorial changes. Permit Condition 2.12 was added to include the Subpart H recordkeeping and annual reporting requirement.

No change was made to Tier I Condition 2.15.

Continuous emission monitoring on the MFC Main Stack is required in accordance with 40 CFR 61.93(e), because under normal operations (but without considering attenuation from the HEPA filters) the potential emissions could cause an effective dose equivalent in excess of 1% of the standard (i.e., 0.1 mrem/yr).

PTC Conditions 2.4 and 2.9

The facility requested deletion of previous permit conditions that limited processing to 90 fuel assemblies per year with average burnup of 10%, and ten fuel assemblies in any 30-day period with average burnup of 15%. These limits were developed for the initial permit issued in 1989, based on the analysis supplied in the application for processing only EBR-II spent fuel. Correspondence dating from that time indicates that the facility was aware that processing other types of fuel (e.g., from FFTF) would likely require a modification to the permit. The revised permit issued on May 9, 2001 deleted the restriction to process only EBR-II fuels, but no analysis was conducted to justify this change.

As a replacement for existing permit conditions 3.2 and 3.3, PTC Condition 2.4 limits annual fuel processing to the 5,000 kilograms per year (11,023 pounds per year) described in the 2000 FEIS, with one change. The emissions inventory and modeled effective dose equivalent calculated in the FEIS for processing the spent fuel at the MFC FCF were based on processing a maximum of 5,000 kilograms per year of driver and blanket fuel, with the amount of driver fuel limited to a maximum of 600 kilograms per year (1,320 pounds per year). To allow greater flexibility in processing the fuels, the facility requested a processing limit of 5,000 kilograms per year of any combination of fuel types. DEQ determined that this appeared to be reasonable based on the following:

a) The effective equivalent dose associated with processing 5,000 kilograms per year of any type of fuel may be greater than 3.4E-04 mrem/yr (the highest EDE estimated for any year in the projected 12-year project), but would be less than 1.98E-03 mrem/yr (the total estimated EDE for processing the DOE entire inventory of sodium-bonded nuclear fuels).

b) At the upper limit of 1.98E-03 mrem/yr, the contribution from these emissions could be expected to constitute between about 2.5% and 25% of the total INL annual dose, which ranged between 7.92E-03 mrem/yr and 7.7E-02 mrem/yr for the years 1999 through 2005.

c) At the upper limit of 1.98E-03 mrem/yr, the contribution from these emissions would be about 0.02% of the maximum 10 mrem dose limit applicable to DOE activities at the INL.

Compliance is demonstrated through the monitoring and recordkeeping requirement contained in Permit Condition 2.9.

PTC Condition 2.8

The HEPA filter conditions were revised to be consistent with those included in other INL permits. For consistency with the existing Tier I permit, note that the HEPA filter conditions were not added to the Tier I operating permit because they are “state-only requirements” (i.e., they address 40 CFR 61 Subpart H requirements and these requirements are not part of the SIP).

Deleted Existing Permit Conditions

- Existing permit conditions 3.2 and 3.3, which limited the number of fuel assemblies and burnup levels.

- Existing permit conditions 5.1, 5.2, and Appendix A, Conditions 3.1 through 3.2.3, which required reporting to DEQ of exceedances of process rates, filter test results within 30 days, and quarterly reports to DEQ documenting that all requirements for HEPA filters had been met.
- A condition suggested by the permittee to require monthly air filter monitoring and laboratory analysis of the MFC stack emissions was not incorporated. DEQ determined that installation, maintenance, and monitoring of the stack CEM in accordance with 40 CFR Subpart H, and maintaining HEPA filters in accordance with an O&M manual and QAPjP was sufficient.

6. PERMIT FEES

A PTC application fee of $1,000.00 applies in accordance with IDAPA 58.01.01 224, and this fee was received on March 30, 2007. A PTC processing fee of $1,000.00 applies in accordance with IDAPA 58.01.01.225, since the potential emissions increase for this new source is less than one ton per year, and the PTC required engineering analysis. [For final, note when the processing fee is received]. Since this is a major facility, Tier I fees are also applicable. As of May 2, 2007, the INL is current with the Tier I fees.

7. PERMIT REVIEW

7.1 Regional Review of Draft Permit

The draft PTC and Statement of Basis were provided to the Idaho Falls Regional Office on April 9, 2007 for review. Replies were received on April 9 (IFRO/AQ) and April 11 (IFRO/INL Oversight). No changes were recommended.

7.2 Facility Review of Draft Permit

The draft PTC and Statement of Basis were provided to DOE-ID and BEA for review on April 13, 2007, and DEQ received minor comments on April 25, 2007.

7.3 Public Comment

In accordance with IDAPA 58.01.01.209.05(c) and 364, a 30-day comment period will be provided for the public, affected states and tribes on the draft PTC and the Tier I operating permit amendment.

IDAPA 58.01.01.008.01 defines affected states as: “All states: whose air quality may be affected by the emissions of the Tier I source and that are contiguous to Idaho; or that are within 50 miles of the Tier I source.” A review of the site location information included in the permit application indicates that the facility is located within 50 miles of Montana and within 50 miles of tribal land. Therefore, Montana and the Shoshone-Bannock Tribes at the Fort Hall Indian Reservation will be provided an opportunity to comment on the draft PTC and Tier I operating permit amendment.

The EPA will also be provided with an opportunity to comment on the proposed Tier I amendment, and this will occur concurrently with the 30-day comment period in accordance with IDAPA 58.01.01.209.05.c.iv and 366.

8. RECOMMENDATION

Based on review of application materials, and all applicable state and federal rules and regulations, staff recommend that DOE-ID be issued a proposed permit for public comment PTC No. 2007.0043 for the MFC FCF. A public comment period will be required, as is review by affected states and the EPA, but the project does not involve PSD requirements.

Appendix A

Excerpts from DOE/EIS-0306:

Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Chapter 2, Proposed Action & Alternatives (Layout Drawing);

Appendix D, Sodium-Bonded Fuel Characteristics; and

Appendix E, Evaluation of Human Health Effects from Normal Operations

P-2007.0043

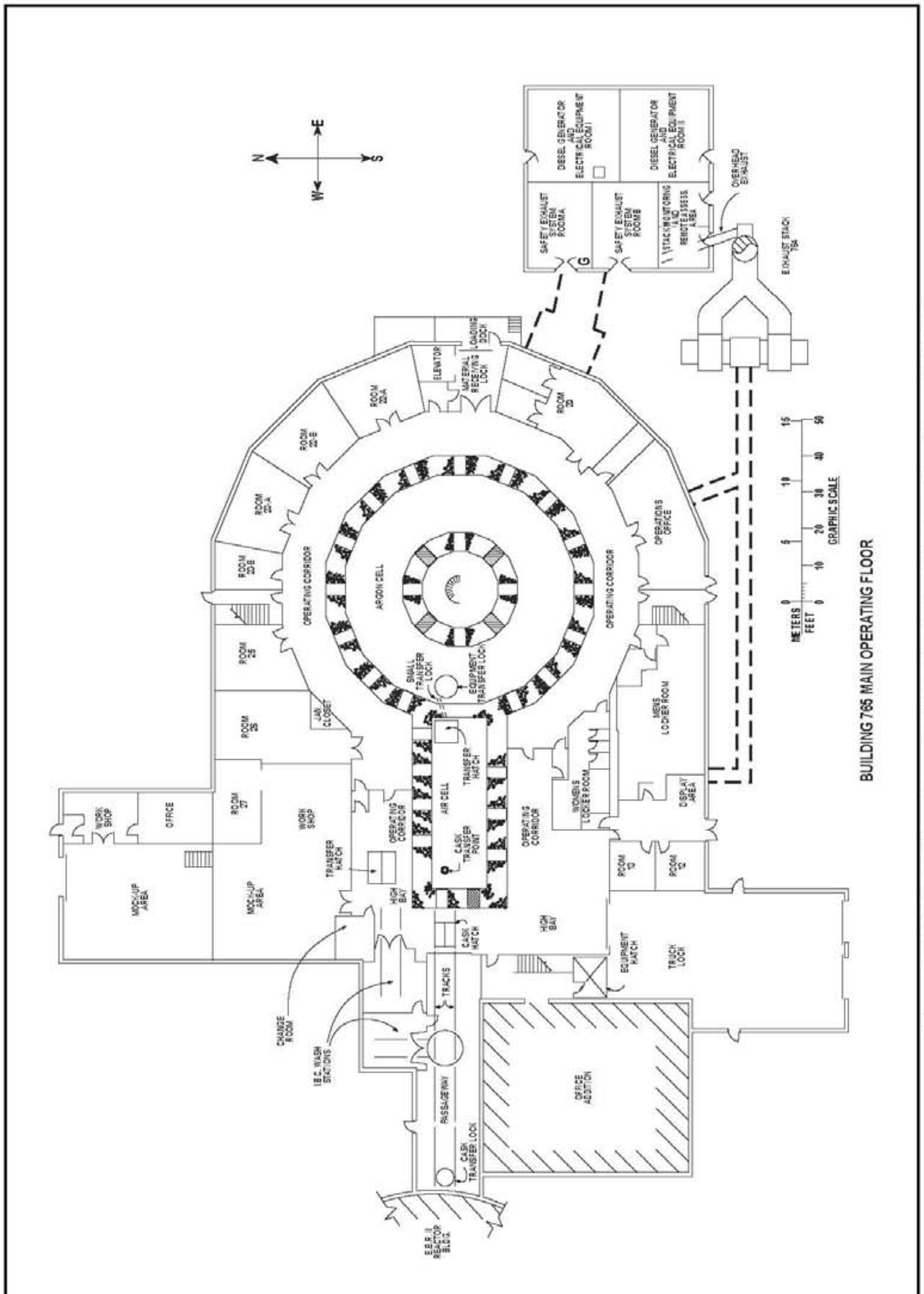

Figure 2–12 Main Floor Layout of the Fuel Conditioning Facility at ANL-W

that were no longer based on the production of strategic nuclear material. DOE identified the initial components of this plan in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995) (hereafter referred to as the Programmatic Spent Nuclear Fuel EIS). The Record of Decision for this EIS (60 FR 28680) stated, in part, that DOE would consolidate the management of its aluminum-clad spent nuclear fuel at the Savannah River Site (SRS), leave the Hanford production spent nuclear fuel at Hanford, and would consolidate nonaluminum-clad fuel at the Idaho National Engineering and Environmental Laboratory (INEEL). This Record of Decision was amended in March 1996 (61 FR 9441). The amended Record of Decision leaves all Fort St. Vrain spent nuclear fuel at the storage site in Colorado, all but sodium-bonded spent nuclear fuel at Hanford, and places restrictions on shipment schedules.

However, in the Programmatic Spent Nuclear Fuel EIS Record of Decision, DOE made no decisions on the technologies it would apply to the management of spent nuclear fuel at the designated storage sites. The Record of Decision stated that the selection of spent nuclear fuel stabilization technologies and the preparation of spent nuclear fuel for ultimate disposition would be the subject of site-specific and fuel-type-specific evaluations prepared in accordance with the National Environmental Policy Act (NEPA) and tiered from the Programmatic Spent Nuclear Fuel EIS (DOE 1995).

D.2 INVENTORY OVERVIEW

This EIS addresses a variety of spent nuclear fuel types that have one common characteristic, the presence of metallic sodium (or sodium and potassium). As a result of research, development, and demonstration activities associated with liquid metal fast breeder reactors, DOE has approximately 60 metric tons of heavy metal of spent nuclear fuel that contains metallic sodium. This EIS addresses a range of technologies that may be used to treat and manage this spent nuclear fuel for disposal. Based on composition, there are five broad categories of spent nuclear fuel to be considered: EBR-II driver spent nuclear fuel, EBR-II blanket, Fermi-1 blanket, Fast Flux Test Facility fuel, and miscellaneous spent nuclear fuel. While there are variations within each category, they may generally be described as follows:

EBR-II driver – This spent nuclear fuel is stainless steel clad highly enriched uranium in a uranium alloy, typically either fissium or zirconium. There are some variations in the specific cladding alloys, the enrichments, fuel compound alloy, dimensions, and burnup within this category. Also, there are small amounts of fuel experiments that use a different uranium compound, for example uranium carbide. This uranium carbide fuel type was added to the miscellaneous group.

EBR-II blanket – This spent nuclear fuel consists of stainless steel clad depleted uranium in a uranium metal form. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary difference between these blankets is dimension and burnup.

Fermi-1 blanket – This spent nuclear fuel consists of stainless steel clad depleted uranium in a uranium-molybdenum alloy. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary difference between these blankets is dimension, elements per assembly, and burnup. Fermi-1 blankets are similar to EBR-II blankets in enrichment, but differ in dimension (Fermi-1 elements are larger), burnup, and form (uranium metal versus uranium-molybdenum alloy).

Fast Flux Test Facility – This group of fuel includes both irradiated and fresh driver fuel. The fuel is either uranium zirconium or plutonium/uranium zirconium, with some containing plutonium/uranium carbide and nitride. This fuel is stainless steel-clad with various levels of enrichment.

Miscellaneous – This group includes experimental spent nuclear fuel from experiments irradiated in the Engineering Test Reactor and the Annular Core Research Reactor at Sandia National Laboratories/New Mexico, Oak Ridge National Laboratory fast reactor spent nuclear fuel, sodium research experiment spent nuclear fuel at SRS, and Westinghouse Atomic Power Division test reactor experiment at INEEL. There are small quantities of experimental fuel that have metallic sodium or potassium. This type of fuel is highly diverse and differs in cladding, uranium compound, enrichment, and burnup.

Table D–1 provides a summary of all DOE sodium-bonded spent nuclear fuel. It should be noted that the inventories reported in Table D–1 include 0.4 metric tons of heavy metal of EBR-II driver fuel and the 1.2 metric tons of EBR-II blanket fuel that are being treated as part of the electrometallurgical treatment demonstration program.

Table D–1 Overview of Sodium-Bonded Spent Nuclear Fuel Categories

Fuel Type	*Storage Volume (cubic meters)* [a]	*Total End of Life Fissile Mass (kilograms)*	*End of Life Mass Metric Tons of Heavy Metal*
EBR-II Driver	58 [b]	2,030	3.1
EBR-II Blanket	13	285	22.4
Fermi-1 Blanket	19	130	34.2
Fast Flux Test Facility	8 [b]	175	0.3
Miscellaneous	3 [b]	60	0.1
Total	**101**	**2,680**	**60**

[a] Volume refers to canister storage volume.
[b] A larger volume per unit mass for the driver fuel is required for the criticality control.
Source: ANL 1999.

By any measure, the majority of the spent nuclear fuel consists of EBR-II driver, EBR-II blanket, and Fermi-1 blanket fuel. **Table D–2** provides a summary of the fraction of spent nuclear fuel in each category by a variety of different measures. As shown, the percentages vary considerably depending upon the measure used for comparison.

Table D–2 Comparison of Sodium-Bonded Spent Nuclear Fuel by Different Measures

Fuel Type	*Storage Volume (percent)*	*Total End of Life Fissile Mass (percent)*	*End of Life Mass Metric Tons of Heavy Metal (percent)*
EBR-II Driver	58	75	5
EBR-II Blanket	13	11	37
Fermi-1 Blanket	19	5	57
Fast Flux Test Facility	8	7	0.5
Miscellaneous	3	2	less than 0.1
Total [a]	**100**	**100**	**100**

[a] Values may not add to exactly 100 percent due to rounding.

The radionuclide inventory of the spent nuclear fuel varies widely due to differences in the construction, function and operational history of the spent nuclear fuel. Therefore, radionuclide inventory estimates were developed for EBR-II driver fuel (including a separate estimate for the experimental driver fuel), EBR-II blanket, Fermi blanket, and Fast Flux Test Facility experimental fuel (SAIC 1999). **Table D–3** provides a summary of plutonium and sodium content for each fuel type.

Table D–3 Plutonium and Sodium Content in Sodium-Bonded Fuel

Spent Nuclear Fuel Type	*Plutonium Mass (kilograms)*	*Sodium Mass (kilograms)*
EBR-II Driver	19	83
EBR-II Blanket	250	176
Fermi-1 Blanket	7	365
Fast Flux Test Facility	3	7
Miscellaneous	0.10	31
Total	**279.10**	**662**

Table D–4 provides a list of principal radionuclide isotopes for each of the fuel types.

For each fuel type, principal radionuclide inventories were determined by considering all isotopes that, as a whole, contribute greater than 99.99 percent of the total dose in a case of accidental release. The dose estimates associated with each isotope intake were based on the effective committed dose equivalent factors provided in Federal Regulatory Guidance Report No. 11 (EPA 1988). Next, the list of isotopes was adjusted to include those isotopes with a boiling point less than 1,400° C (2,550° F), which is the maximum melt and dilute process temperature, and then isotopes of interest like hydrogen-3 (tritium), krypton-85, iodine-129, and uranium isotopes were added. The values in Table D–4 reflect the inventory of each isotope as of January 2000 (Liaw 1998).

The following sections provide a more detailed description of each category of spent nuclear fuel.

D.3 EBR-II Spent Nuclear Fuel

D.3.1 Reactor Background

EBR-II was a research and test reactor at Argonne National Laboratory-West (ANL-W) used to demonstrate the engineering feasibility of a sodium-cooled, liquid metal fast breeder reactor with a steam electric power plant and integral fuel cycle. It achieved initial criticality in September 1961 and continued to operate until September 1994. During its operation, numerous fuel designs were tested in EBR-II. The reactor operating power level was 62.5 megawatts-thermal.

D.3.2 Description of EBR-II Spent Nuclear Fuel

The EBR-II reactor consisted of an enriched driver core surround by depleted blanket assemblies. The reactor originally had an upper and lower axial blanket above and below the driver core, as well as a radial blanket around the perimeter of the driver core. It later operated with a radial blanket only. In addition, various experimental assemblies where placed into the core for testing. The following sections describe the driver fuel (including experiments) and blanket assemblies.

Table D–4 Principal Radionuclide Activities per Kilogram of Heavy Metal [a]

Elements	*Isotope*	*EBR-II Driver* [b]	*EBR-II Radial Blanket* [c]	*EBR-II Exp. Driver Fuel*	*Fermi-1 Blanket*	*FFTF Driver*
Tritium	H-3	1.23	0.00712	1.16	0.0000756	1.90
Carbon	C-14	0.000199	0.0000597	0.000954	1.05×10^{-8}	0.000674
Iron	Fe-55	4.87	0.0901	5.11	0.0000269	9.89
Cobalt	Co-60	0.481	0.0159	2.09	0.0000888	0.586
Nickel	Ni-63	0.229	0.00306	0.152	0.0000482	0.0491
Krypton	Kr-85	18.9	0.0520	16.5	0.000663	23.9
Strontium	Sr-90	197	0.807	171	0.0163	241
Yttrium	Y-90	197	0.807	171	0.0163	241
Ruthenium	Ru-106	1.51	0.135	2.67	7.02×10^{-10}	3.95
Rhodium	Rh-106	1.51	0.135	2.67	7.02×10^{-10}	3.95
Cadmium	Cd-113M	0.0464	0.000712	0.0511	2.86×10^{-6}	0.0659
Antimony	Sb-125	2.96	0.0231	2.98	2.92×10^{-6}	4.72
Tellurium	Te-125M	1.23	0.00951	1.23	1.20×10^{-6}	1.89
Iodine	I-129	0.0000735	1.44×10^{-6}	0.0000685	1.26×10^{-8}	0.0000898
Cesium	Cs-134	1.76	0.0134	1.93	6.66×10^{-9}	4.19
	Cs-137	221	1.73	199	0.0243	272
Barium	Ba-137M	209	1.64	188	0.0230	257
Cerium	Ce-144	2.96	0.0627	5.55	6.60×10^{-12}	9.88
Praseodymium	Pr-144	2.96	0.0627	5.55	6.60×10^{-12}	9.88
Promethium	Pm-147	82.6	0.407	80.2	0.0000810	128
Samarium	Sm-151	5.34	0.100	5.00	0.00131	6.49
Europium	Eu-154	0.567	0.00734	0.628	7.70×10^{-7}	0.969
	Eu-155	3.81	0.0481	3.97	0.0000671	5.28
Thorium	Th-228	0.0000514	1.55×10^{-7}	0.0000561	1.32×10^{-10}	0.0000739
Uranium	U-234	0.0404	1.33×10^{-6}	0.0371	3.20×10^{-8}	0.0407
	U-235	0.00131	3.77×10^{-6}	0.00120	7.48×10^{-6}	0.00123
	U-236	0.00121	4.24×10^{-6}	0.00104	1.09×10^{-7}	0.00141
	U-238	0.000111	0.000327	0.000120	0.000331	0.000117
Neptunium	Np-237	0.000289	8.37×10^{-6}	0.000287	2.28×10^{-7}	0.000401
Plutonium	Pu-238	0.166	0.00939	0.233	3.34×10^{-6}	0.304
	Pu-239	0.269	0.753	1.61	0.0134	0.739
	Pu-240	0.00911	0.0518	0.754	0.0000112	0.123
	Pu-241	0.00222	0.210	14.4	3.54×10^{-7}	1.60
Americium	Am-241	0.000391	0.0163	0.359	3.46×10^{-8}	0.0516
Americium	Am-242M	3.313×10^{-7}	0.000169	0.00218	7.84×10^{-14}	0.000140
Total	**Ci/kg** [d]	**957**	**7.18**	**884.1**	**0.0959**	**1,240**
Total heavy metal mass	metric tons	3.1	22.4	0.2 [e]	34.2	0.25

[a] Activities are in curies per kilogram of heavy metal, as of January 1, 2000.
[b] Inventory of Mark III driver fuel is bounding fuel for all EBR-II driver fuel type.
[c] Representative for all EBR-II blanket fuel.
[d] Curie per kilogram of heavy metal.
[e] EBR-II experimental driver fuel mass is a subset of EBR-II driver fuel.

Table E–5 Annual and Total Radiological Releases During Normal Operations Under Alternative 1 at ANL-W

	Annual Releases (curies per year)			
Isotope [a]	*Years 1 through 5*	*Year 6*	*Years 7 through 12*	*Project Lifetime Total (curies)*
H-3	770	680	0.38	4,530
C-14	1.7×10^{-12}	1.0×10^{-12}	2.3×10^{-16}	9.4×10^{-12}
Fe-55	1.4×10^{-8}	1.5×10^{-8}	5.8×10^{-13}	8.7×10^{-8}
Co-60	1.6×10^{-9}	9.7×10^{-10}	1.9×10^{-12}	8.8×10^{-9}
Ni-63	6.5×10^{-10}	1.7×10^{-10}	1.0×10^{-12}	3.4×10^{-9}
Kr-85	11,570	8,800	3.3	66,670
Sr-90	7.0×10^{-8}	5.2×10^{-8}	4.7×10^{-11}	4.0×10^{-7}
Y-90	7.0×10^{-8}	5.2×10^{-8}	4.7×10^{-11}	4.0×10^{-7}
Ru-106	3.2×10^{-8}	2.9×10^{-8}	7.6×10^{-17}	1.9×10^{-7}
Rh-106	3.2×10^{-8}	2.9×10^{-8}	7.6×10^{-17}	1.9×10^{-7}
Cd-113m	6.7×10^{-10}	5.2×10^{-10}	3.1×10^{-13}	3.9×10^{-9}
Sb-125	4.1×10^{-8}	3.6×10^{-8}	3.2×10^{-13}	2.4×10^{-7}
Te-125m	4.5×10^{-10}	3.9×10^{-10}	3.4×10^{-15}	2.6×10^{-9}
I-129	1.4×10^{-12}	9.7×10^{-13}	1.8×10^{-15}	8.2×10^{-12}
Cs-134	3.2×10^{-8}	4.0×10^{-8}	9.5×10^{-16}	2.0×10^{-7}
Cs-137	4.0×10^{-6}	2.9×10^{-6}	3.5×10^{-9}	0.000023
Ba-137m	3.8×10^{-6}	2.8×10^{-6}	3.3×10^{-9}	0.000022
Ce-144	1.2×10^{-9}	1.8×10^{-9}	1.9×10^{-20}	7.7×10^{-9}
Pr-144	1.2×10^{-9}	1.8×10^{-9}	1.9×10^{-20}	7.7×10^{-9}
Pm-147	2.9×10^{-8}	2.6×10^{-8}	2.3×10^{-13}	1.7×10^{-7}
Sm-151	2.1×10^{-9}	1.4×10^{-9}	3.7×10^{-12}	1.2×10^{-8}
Eu-154	2.1×10^{-10}	2.0×10^{-10}	2.2×10^{-15}	1.3×10^{-9}
Eu-155	1.4×10^{-9}	1.1×10^{-9}	1.9×10^{-13}	8.3×10^{-9}
Th-228	1.6×10^{-14}	1.3×10^{-14}	3.2×10^{-19}	9.1×10^{-14}
U-234	1.2×10^{-11}	7.8×10^{-12}	7.8×10^{-17}	6.7×10^{-11}
U-235	3.9×10^{-13}	2.6×10^{-13}	1.8×10^{-14}	2.3×10^{-12}
U-236	3.7×10^{-13}	2.6×10^{-13}	2.7×10^{-16}	2.1×10^{-12}
U-238	7.4×10^{-13}	7.7×10^{-13}	8.1×10^{-13}	9.4×10^{-12}
Np-237	3.9×10^{-13}	2.8×10^{-13}	2.1×10^{-15}	2.2×10^{-12}
Pu-238	2.9×10^{-10}	2.2×10^{-10}	3.4×10^{-14}	1.6×10^{-9}
Pu-239	7.1×10^{-9}	1.2×10^{-9}	1.4×10^{-10}	3.7×10^{-8}
Pu-240	4.7×10^{-10}	1.2×10^{-10}	1.1×10^{-13}	2.5×10^{-9}
Pu-241	1.9×10^{-9}	1.1×10^{-9}	3.6×10^{-15}	1.1×10^{-8}
Am-241	6.2×10^{-12}	1.8×10^{-12}	1.5×10^{-17}	3.3×10^{-11}
Am-242m	6.4×10^{-14}	9.3×10^{-15}	3.4×10^{-23}	3.3×10^{-13}
Totals	12,310	9,500	3.7	71,200

[a] The listed isotopes are present within the argon cell at the Fuel Conditioning Facility. Due to lack (scarcity) of oxygen in the argon cell, the tritium (H-3) released to the cell would be in molecular (elemental) form.

Population Impacts

The estimated annual radiological impacts due to the source term for the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding ANL-W are tabulated in **Table** E–6. Calculated impacts are shown for each year of processing as well as for each of the fuel types to be processed. Impacts are listed resulting from releases during processing EBR-II driver and blanket spent nuclear fuel during each of the first five years (years 1 through 5), processing some of all four fuel types during the sixth year (year 6), and processing Fermi-1 blanket spent nuclear fuel during each of the final six years (years 7 through 12). The impacts to the maximally exposed offsite individual and the surrounding population would result primarily from estimated releases of tritium (H-3) and krypton-85. Together, these two radionuclides would account for greater than 99.9 percent of the estimated impacts.

Table E–6 Annual Radiological Impacts to the Public From Operational Activities Under Alternative 1 at ANL-W

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Processing	*Spent Nuclear Fuel Type*	*Collective Dose (person-rem per year)*	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem per year)*	*Latent Cancer Fatality Risk*
1 - 5	EBR-II driver	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
	Fast Flux Test Facility driver	0	0	0	0
	EBR-II blanket	0.000083	4.2×10^{-8}	0.000010	5.0×10^{-12}
	Fermi-1 blanket	0	0	0	0
	All fuel, years 1 through 5	0.0028	1.4×10^{-6}	0.00034	1.7×10^{-10}
6	EBR-II driver	0.00046	2.3×10^{-7}	0.000054	2.7×10^{-11}
	Fast Flux Test Facility driver	0.0018	9.2×10^{-7}	0.00022	1.1×10^{-10}
	EBR-II blanket	7.6×10^{-6}	3.8×10^{-9}	9.1×10^{-7}	4.6×10^{-13}
	Fermi-1 blanket	9.1×10^{-7}	4.5×10^{-10}	1.1×10^{-7}	5.5×10^{-14}
	All fuel, year 6	0.0023	1.2×10^{-6}	0.00028	1.4×10^{-10}
7 - 12	EBR-II driver	0	0	0	0
	Fast Flux Test Facility driver	0	0	0	0
	EBR-II blanket	0	0	0	0
	Fermi-1 blanket	1.1×10^{-6}	5.4×10^{-10}	1.3×10^{-7}	6.5×10^{-14}
	All fuel, years 7 through 12	1.1×10^{-6}	5.4×10^{-10}	1.3×10^{-7}	6.5×10^{-14}

Total cumulative radiological impacts over the projected 13 years of operations under this alternative are tabulated in **Table** E–7. This table shows the sum of the calculated impacts to the maximally exposed offsite individual and the surrounding population over 12 years of fuel treatment.

Table E–7 Cumulative Maximum Radiological Impacts to the Public From Normal Operational Releases Under Alternative 1 at ANL-W

	Population		*Maximally Exposed Offsite Individual*	
	Collective Dose (person-rem)	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem)*	*Latent Cancer Fatality Risk*
Project total impacts [a]	0.0163	8.2×10^{-6}	0.00198	9.9×10^{-10}

[a] Total impacts are estimated for the 12-year duration of fuel treatment; there are no releases in the 13th year, i.e., only salt stabilization is performed.

Worker Impacts

Workers involved with electrometallurgical treatment activities at ANL-W could receive radiation doses during handling activities, such as receiving and unloading fuel casks, and transferring in-process waste material from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. Doses received during in-cell activities likely would be very small. A maximally exposed worker dose estimate for this EIS is based on the regulatory limit of 5,000 millirem per year for radiation workers at DOE sites. If an individual worker received this dose each year of the 13 years of the electrometallurgical treatment project, the total worker dose would be 65,000 millirem with an associated risk of developing fatal cancer of 0.026.

However, actual worker doses are likely to be much lower than this maximum estimate. The ANL-W radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. In addition, ANL-W has established an administrative goal of 1,500 millirem per year to any individual. The general design goals at the Fuel Conditioning Facility, for example, were to maintain radiation fields below 0.5 millirem per hour at all workstations. This means that for an individual working at the Fuel Conditioning Facility for a full-time occupational work year of 2,000 hours, the annual dose would be 1,000 millirem.

Worker population doses were estimated by examining the type and duration of various operations performed by workers involved with the electrometallurgical treatment project. Doses can be estimated based on previous doses from similar activities at ANL-W. Based on information from ANL-W, the total worker population dose estimate is 22 person-rem per year, averaging out to an individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates are extended out over the 13 years of operational activities (12 years of fuel treatment and a year of high-level radioactive waste conversion activities), the collective worker dose is 286 person-rem and the associated risk is 0.11 latent cancer fatalities. The estimated impacts to the worker population associated with this alternative are summarized in **Table E–8**.

Table E–8 Annual and Total Impacts to Workers From Operational Activities Under Alternative 1 at ANL-W

	Worker Population	
	Collective Dose (person-rem)	*Latent Cancer Fatalities*
Annual impacts	22	0.0088
Project total impacts [a]	319	0.13

[a] Total impacts are estimated for the 13-year processing duration, plus a year for deactivation activities at 33 person-rem.